ION

THIS

MET

Conf- 820802 --14

LA-UR--82-2298

DE82 021922

MASTER

TITLE: COMPARISONS OF TRAC-PF1 CALCULATIONS WITH SEMISCALE MOD-3
SMALL-BREAK TESTS S-07-10D, S-SB-P1, AND S-SB-P7

AUTHOR(S): Manjit S. Sahota, Q-9

Los Alamos
Los Alamos National Laboratory
Los Alamos, New Mexico 87545

COMPARISONS OF TRAC-PF1 CALCULATIONS WITH SEMISCALE MOD-3
SMALL-BREAK TESTS S-07-10D, S-SB-P1, AND S-SB-P7*

M. S. Sahota

Safety Code Development
Energy Division
Los Alamos National Laboratory
Los Alamos, New Mexico 87545, U.S.A.

ABSTRACT

Semiscale Tests S-07-10D, S-SB-P1, and S-SB-P7 conducted in the Semi-scale Mod-3 facility at the Iduho National Engineering Laboratory are analyzed using the latest released version of the Transient Reactor Analy-sis Code (TRAC-PF1). The results are used to assess TRAC-PF1 predictions of thermal-hydraulic phenomena and the effects of break size and pump oper-ation on system response during slow transients. Test S-07-10D simulated an equivalent pressurized-water-reactor (PWR) 10% communicative cold-leg break for an early pump trip with an emergency core coolant (ECC) injected only into the intact-loop cold leg. Tests S-SB-P1 and S-SB-P7 simulated 2.5% communicative cold-leg breaks for early and late pump trips, respec-tively, with only high-pressure injection (HPI) into the cold legs. The parameters examined include break flow, primary-system pressure response, primary-system mass distribution, and core characteristics. For Test S-07-10D, the calculated core uncovery began ~100 s earlier than the measured uncovery. The calculated peak cladding temperature was ~100 K less than that in the data because of faster system depressurization, which was responsible for the earlier ECC injection. For Test S-SB-P1, the ex-perimental core uncovery began at ~800 s into the transient. The base-case calculation showed that the core was on the verge of uncovering after ~600 s, but no distinct core uncovery was predicted. However, when the break flow was increased by ~10% (significantly within the uncertainty of the experimental data), a core uncovery similar to that in the data was calculated. For Test S-SB-P7, the core uncovery was neither observed nor calculated.

INTRODUCTION

The Transient Reactor Analysis Code (TRAC) is an advanced best-estimate systems code for analyzing postulated accidents in light-water reactors. The latest released

---

version of the code (TRAC-PF1) [1] provides this analysis capability for pressurized-water reactors (PWRs) and for a wide variety of thermal-hydraulic experimental facilities.

Semiscale Tests S-07-10D, S-SB-P1 and S-SB-P7 [2,3] were conducted in the Semiscale Mod-3 facility at the Idaho National Engineering Laboratory (INEL) to investigate the thermal-hydraulic phenomena resulting from a communicative small-break loss-of-coolant accident (LOCA) in a PWR. The primary factors differentiating the tests are the break size and the operation of the primary-coolant pumps. The resulting data are used to assess the analytical capability of TRAC-PF1. Of particular interest are the effects of break size and primary-coolant pump operation on the core thermal response. Effects associated with the emergency-core-coolant (ECC) injection, steam-generator heat transfer, slab and rod heat transfer, and break-flow model also are investigated.

SEMISCALE MOD-3 SYSTEM DESCRIPTION

The Semiscale Mod-3 system is a small-scale model of a four-loop PWR and includes an intact loop, a broken loop, an external downcomer assembly, and a pressure vessel. The intact loop includes a pressurizer, steam generator, and pump. The broken loop includes a steam generator, pump, and rupture valve assembly. The pressure vessel includes an upper head, an upper plenum, a 25-rod electrically heated core with thermocouples located 0.75 mm beneath the cladding surface, and a lower plenum. The external downcomer assembly includes an inlet annulus and downcomer pipe. Most system components have the same elevations as those in a full-sized PWR. The Semiscale Mod-3 system design description [4] contains additional details on the Mod-3 system.

TEST DESCRIPTIONS

Test S-07-10D was performed to characterize experimentally the thermal-hydraulic behavior of the Mod-3 system. The test simulated a 10% cold-leg communicative break with pump coastdowns beginning early in the transient (2.6 s after the pressurizer pressure reached 12.41 MPa). The simulated core consisted of 9 high-power rods (46.7 kW/m average), 13 low-power rods (30.9 kW/m average), and 3 unpowered rods in a 5 × 5 matrix. The initiation of the ECC injection was delayed until the 1060-K peak rod temperature was attained. The ECC was injected only into the intact loop. The secondary side of the broken-loop steam generator was blown down through the steam discharge valve during the entire transient to examine the influence of the secondary-side conditions on primary-side behavior.

Tests S-SB-P1 and S-SB-P7 simulated 2.5% cold-leg communicative breaks with pump coastdowns beginning early and late (3.4 s and 1099.7 s, respectively, after the pressurizer pressure reached 12.48 MPa). The simulated core had a flat radial power profile with three unpowered rods in the matrix. Core power decay, pump coastdowns, and steam-generator valve actions were sequenced relative to a trip signal generated by a specified low pressure (12.48 MPa) in the pressurizer. The ECC was provided by the high-pressure injection system (HPIS) only. The accumulators in the intact and broken loops were valved out and the test was terminated before the system pressure fell below the normal low-pressure injection system (LPIS) set point.

For Tests S-SB-P1 and S-SB-P7 the pressure-suppression tank was bypassed and the break discharge was drained through a condensing system into a small catch tank. The catch tank inventory was measured before and after the test to obtain the total integrated break flow.

TRAC MODEL

The TRAC input model for the Semiscale Mod-3 facility generally corresponds to the hardware configuration. Although TRAC-PF1 can model a three-dimensional vessel, all vessel elements are modeled using one-dimensional components to assess their utility and to save computation time. The TRAC-PF1 choked-flow model is used to calculate the break flow. The input model consists of 42 components containing a total of 171 computational cells for Test S-07-10D and 172 computational cells for Tests S-SB-P1 and S-SB-P7.

RESULTS

Test S-07-10D (10% Break with Early Pump Trip)

Figure 1 compares the experimental and calculated break flows. The agreement is good with the calculated flow occurring mostly within the data scatter. The sharp spikes in the measured break flow at ~50 s may be caused by flashing in the intact-loop steam generator that forces fluid through the broken-loop hot leg to the break. The sharp spikes in the calculation at ~360 s are caused by the spikes in the fluid density upstream of the break resulting from accumulator injection.
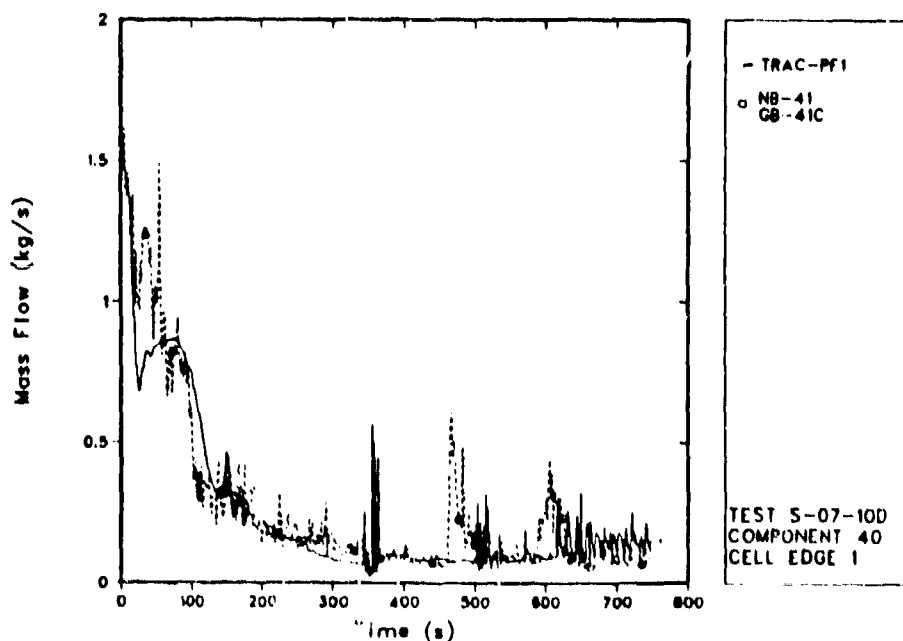

Fig. 1.
Break flows for Semiscale Test S-07-10D.

Figure 2 shows the calculated and measured upper-plenum pressures. The calcula-
ted pressure drops at a faster rate than the data between 200 and 370 s. Because the
ECC trips are based on system pressure, the ECC injection sequence in the calculation
precedes that in the data by 122 s and the accumulator injection begins at 336 s. A
sharp pressure increase after the accumulator injection is caused by core quench that
increases the vapor generation rate. The calculation corresponds to the data except
for a time delay after the accumulator injection.

Generally, the calculated liquid distribution in the system compares well with
the distribution in the experiment (the liquid masses were estimated from fluid
densities) with the following exceptions.

1. The broken-loop hot leg in the experiment is, on the
    average, ~30% full of liquid between 100 and 400 s whereas
    the calculation shows almost no liquid. However, the
    broken-loop hot-leg liquid volume is only ~1% of the total
    primary-system volume. Thus, this discrepancy does not have
    any noticeable impact on the overall system behavior.

2. The intact-loop pump suction leg remains, on the average,
    ~70% full of liquid up to 300 s in the calculation, whereas
    the experiment shows only ~25% liquid in the leg. The pump
    suction leg volume is ~11% of the total system volume, which
    can be ~17 kg of liquid. Thus, during this time period, the
    calculation shows ~7 kg more liquid in the pump suction leg
    than the experiment. The initial primary-system liquid mass
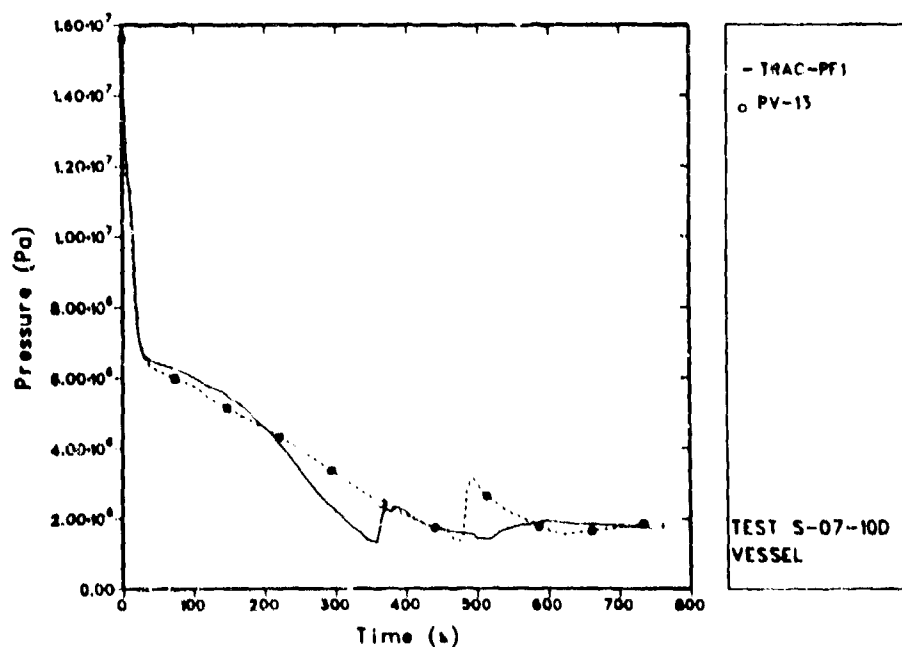   is ~148 kg.

Fig. 2.
Upper-plenum system pressures for Semiscale Test S-07-10D.

3. The broken-loop pump suction leg in the calculation voids at
    ~200 s, whereas the experimental voiding occurs just before
    500 s. The hot-leg liquid volume represents ~3% (~4 kg of
    liquid) of the total primary-system volume. The inaccura-
    cies in the liquid distributions apparently did not
    influence the overall system behavior significantly.

Figure 3 shows the calculated and measured clad temperatures near the center of
the core. The calculated core uncovery begins ~100 s earlier than in the data.
Because the faster system depressurization causes an early ECC trip, the quenching
also starts ~100 s earlier than in the data. This early ECC injection causes the
calculated peak temperature to be lower than the experimental peak temperature.

It took 4567 s of central-processor-unit (CPU) time on a CDC 7600 to simulate a
748-s system transient at an average 0.12-s time step. The running time to simulate
the same length of transient using TRAC-PD2 [5] was 11 376 s.

Test S-SB-P1 (2.5% Break with Early Pump Trip)

Figure 4 shows experimental and calculated system pressure histories. During
the first 1000 s of the transient, the pressure is overpredicted by an average of
~10%. At least a part of this pressure overprediction results from the lower break-
flow prediction (although the transient break-flow data are not available, an ~8%
underprediction in the integrated break flow is estimated from the catch-tank mea-
surement). Also, during the first 1000 s of the transient, the pressure is sensitive
to the system heat loss to the surroundings that has considerable uncertainty.*

The density comparisons in the loops (not illustrated) show, in general, good
comparisons with the data with an average discrepancy of ~100 kg/m^3. Thus, TRAC-PF1
satisfactorily calculates the liquid mass distributions in the loops for Test
S-SB-P1. The calculated liquid mass in the vessel, therefore, should be very close
to that in the data. However, the cladding temperature comparisons show that core
dryout is observed near the top whereas the prediction does not show any such ten-
dency. However, a void fraction of ≥ 0.7 is calculated near the top of the core when
it is supposed to uncover, which indicates that the core is on the verge of
uncovering. The primary reason for this failure to calculate the core uncovery is
the lower break-flow prediction.

To investigate the effect of break flow (which is underpredicted by ~8%) on the
core thermal response, a sensitivity run was made by artificially increasing the
break area to achieve a more accurate break-flow calculation. As a result, the break
flow in this run actually is overpredicted by ~2%. Figure 5 compares the clad tem-
peratures in the upper part of the core for this run. The comparison is excellent
with the core dryout predicted at the right time. The clad temperatures at lower
elevations also are in good agreement with those in the data with no core dryout
predicted at these locations as indicated by the data.

The CDC 7600 CPU time required to run a 1671-s system transient was 2860 s at
an average 0.37-s time step. The running time to simulate the same length of tran-
sient using TRAC-PD2 [5] was 22 136 s.

---

*A primary-system steady-state heat loss of 125 kW was modeled in TRAC. The actual
loss is estimated to be between 80-180 kW [Semiscale Review Group Meeting, presenta-
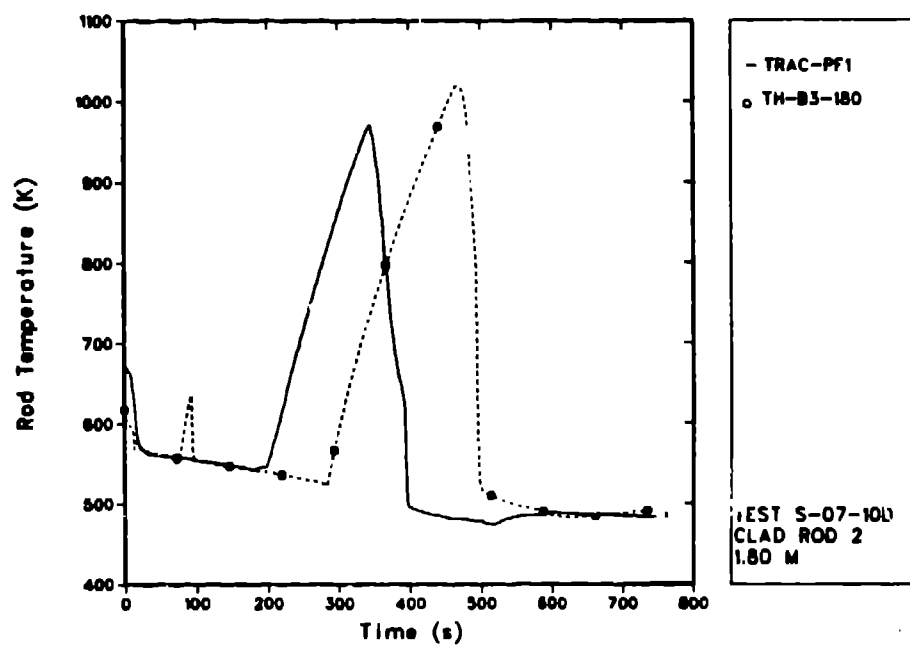tion by A. G. Stephens (August 18, 1981)].

Fig. 3.
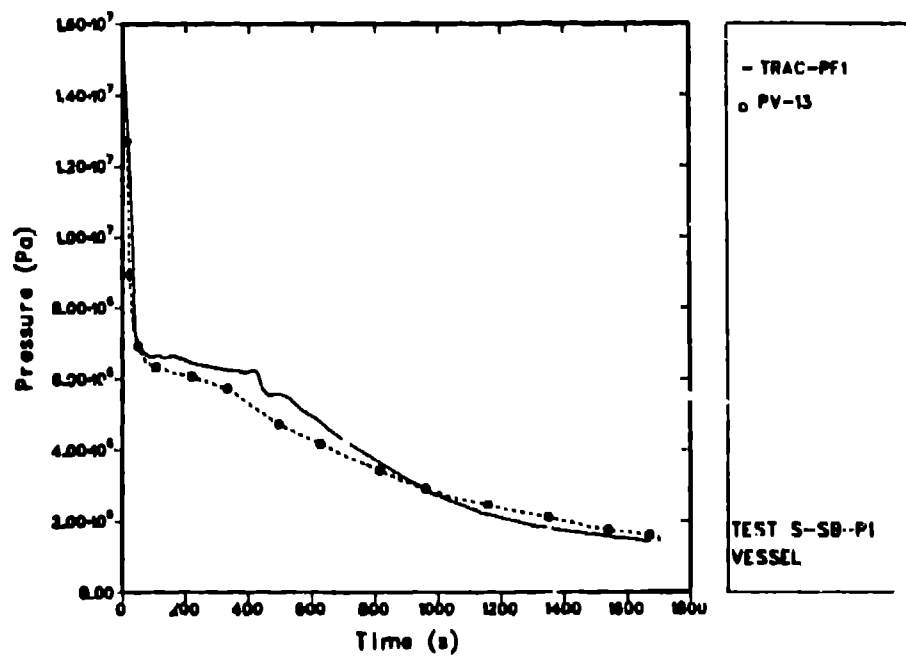Clad temperatures at 1.80-m elevation for Semiscale Test S-07-10D.

Fig. 4.
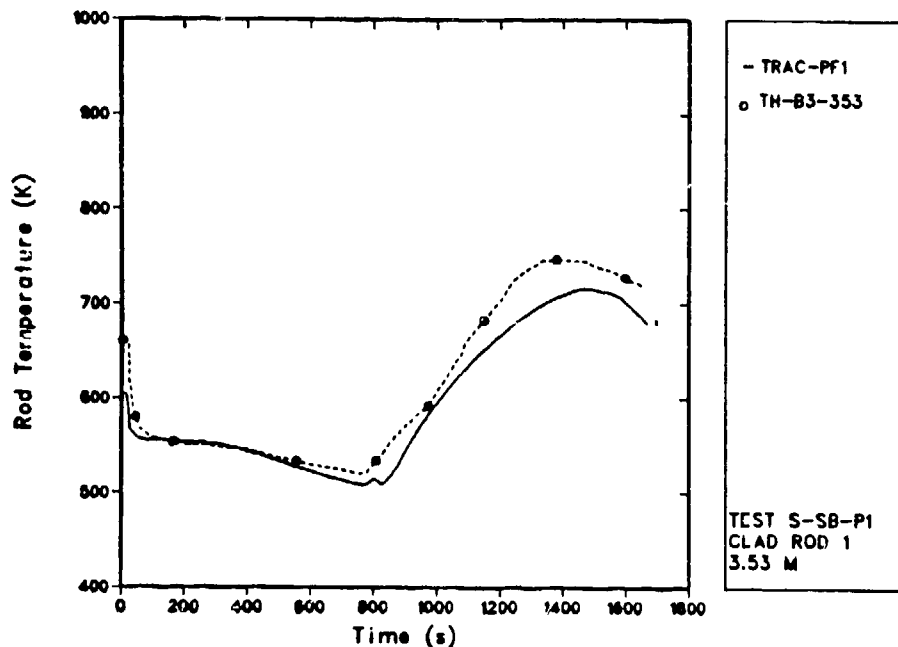Upper-plenum pressures for Semiscale Test S-SB-P1.

Fig. 5.
Comparison of the clad temperatures at the 3.53-m elevation between the experimental data for Semiscale Test S-SB-P1 and the TRAC-PF1 calculation with increased break flow.

Test S-SB-P7 (2.5% Break With Late Pump Trip)

Figure 6 shows the experimental and calculated break flows. The mass flow is overpredicted between 300 and 1000 s of the transient because of a higher density prediction upstream of the break during this time. However, the overprediction in the break flow may not be as large as it appears in Fig. 6 because the instrument reading after 500 s lies mostly in the dead-band range. The measured mass-flow uncertainty, therefore, is expected to be much larger than shown in Fig. 6. A better estimate of the error in the calculated break flow is made by comparing the integrated flows with the catch-tank measurements. Such a comparison shows that the flow is underpredicted by an average of 5% for the first 814.6 s and overpredicted by an average of 29% during the rest of the transient, with an average overprediction of only 4% for the entire transient. This suggests that the actual flow during the first 300 s of the transient must have been significantly larger than indicated by the measurement. These comparisons clearly point to the large uncertainty in the experimental data plotted in Fig. 6.

Figure 7 shows experimental and predicted system pressure histori . The pressure is slightly overpredicted during the first 1000 s and underpredicted during the rest of the transient. The discrepancy in the pressure calculation is caused primarily by the inaccuracy in the break-flow calculation, which is underpredicted during the first one third of the transient and overpredicted during the rest of the transient. The pressure also is sensitive to the system heat loss, as mentioned earlier.
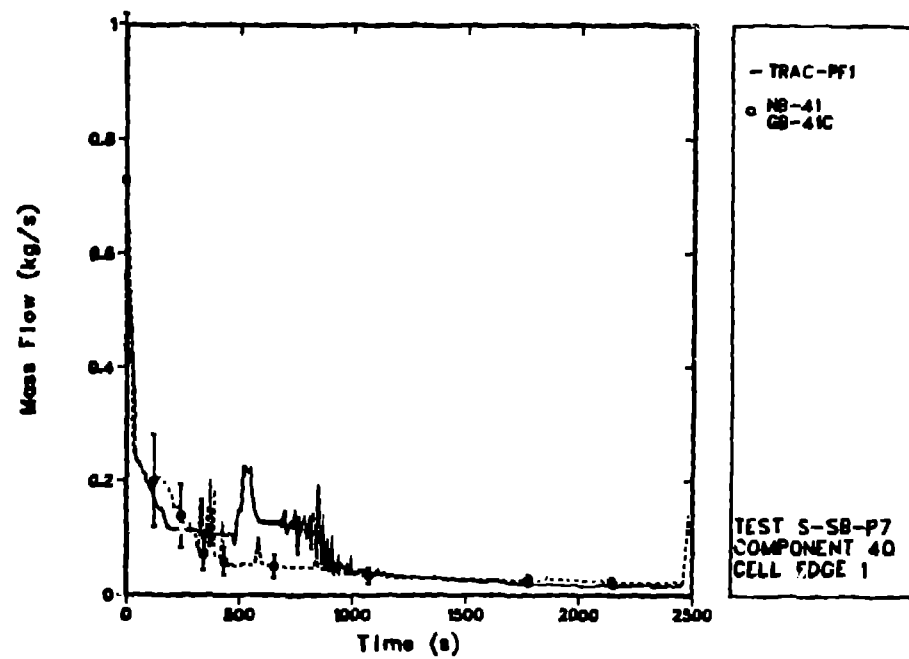
Fig. 6.
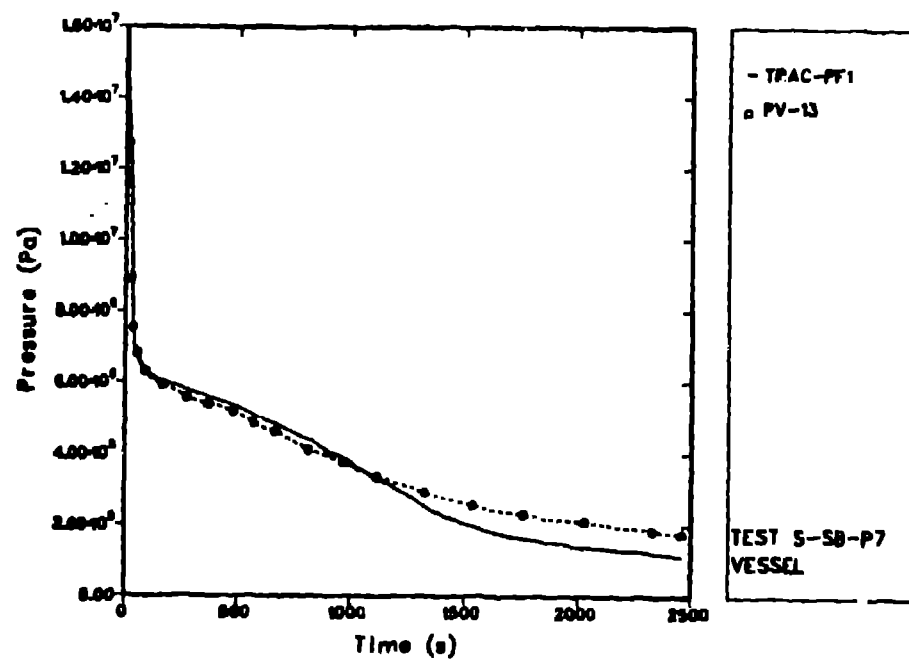Break flows for Semiscale Test S-SB-P7.

Fig. 7.
Upper-plenum pressures for Semiscale Test S-SB-P7.

The calculated density comparisons (not shown), in general, are in good agreement with the data with the exception that during the first 1000 s of the transient the calculated density decays do not occur as rapidly as those in the experiment. This is primarily the result of the lower break-flow prediction during this time. The calculated liquid distribution in the system, therefore, should be approximately the same as that in the experiment.

For Test S-SB-P7 core uncovery is neither observed nor calculated. Thus, the cladding temperatures (not presented) at various elevations in the core are slightly above saturation temperature in both the calculation and the experiment.

It took 5052 s of CPU time on a CDC 7600 to simulate a 2465-s system transient at an average 0.29-s time step. The running time to simulate the same length of transient using TRAC-PD2 [5] was 42 839 s.

CONCLUSIONS

TRAC-PF1 provides a reasonable small-break modeling capability for predicting slow-transient thermal-hydraulic phenomena during a cold-leg break. Most comparisons between TRAC-PF1 results and experimental data generally predict correct trends. This conclusion was made by comparing the break flows, system pressures, primary-side fluid densities, and clad temperatures.

For Test S-07-10D, between 150 and 350 s the calculated system depressurization occurred somewhat faster than the experimental depressurization. Consequently, the calculated ECC injection started 122 s earlier than in the data. This early ECC injection did not allow the calculated peak clad temperature to go as high as that observed in the experiment.

TRAC-PF1 predicts the break flow well within the uncertainty of the measurement. However, more accurate measurement of the transient break flow is highly desirable because some inconsistencies in the transient break flow and the catch-tank measurements have been found.

In both the experiment and the calculation, Test S-SB-P1 with early pump trip was found to be more severe with respect to core thermal response than Test S-SB-P7 with late pump trip.

In conclusion, TRAC-PF1 predicts most of the thermal-hydraulic phenomena resulting from early and late pump-trip small-break LOCAs within the confines of the uncertainty in the boundary conditions. In general, quantitatively good break flows, system pressures, liquid mass distributions, and core thermal response have been calculated. No TRAC-PF1 modeling deficiencies were found. However, if more accurate measurement of the break flow could be achieved, it would be desirable to improve the TRAC choking model.

REFERENCES

1. J. H. MAHAFFY, D. R. LILES, and T. F. BOTT, "TRAC Methods and Models,"
   Proc. Am. Nucl. Soc. Specialists Meeting on Small Break Loss-of-Coolant Accident
   Analyses in LWRs, Monterey, California (August 25-27, 1981), Report WS-81-201,
   Electric Power Research Institute (1981).

2. K. E. SACKETT and L. B. CLEGG, "Experiment Data Report for Semiscale Mod-3 Small
   Break Test S-07-10D (Baseline Test Series)," Report NUREG/CR-1641, EGG-2065,
   Idaho National Engineering Laboratory (December 1980).

3. L. L. WEIDERT and L. B. CLEGG, "Experiment Data Report for Semiscale Mod-3 Small
   Break Test Series (Tests S-SB-P1, S-SB-P2, and S-SB-P7)," Report NUREG/CR-1640,
   EGG-2053, Idaho National Engineering Laboratory (September 1980).

4. "System Design Description for the Mod-3 Semiscale System," Original Issue
   (July 1978), Revision A (June 1979), EG&G Idaho, Inc.

5. "TRAC-PD2, An Advanced Best-Estimate Computer Program for Pressurized Water
   Reactor Loss-of-Coolant Accident Analysis," Report LA-8709-MS, NUREG/CR-2054, Los
   Alamos National Laboratory (April 1981).